NON-PUBLIC?: N
ACCESSION #: 9406160107
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Surry Power Station, Unit 1 PAGE: 1 OF 4

DOCKET NUMBER: 05000280

TITLE: Unit 1 Manual Reactor Trip During Safeguards Actuation
Logic Testing
EVENT DATE: 05/11/94 LER #: 94-006-00 REPORT DATE: 06/09/94

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: M. R. Kansler, Station Manager TELEPHONE: (804) 357-3184

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: JE COMPONENT: CL MANUFACTURER:
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 11, 1994, with Unit 1 initially at 100% power, the reactor was manually tripped at 2109 hours. The trip was initiated when one of the two sources of main feedwater was isolated and Operators were unable to recover steam generator levels. The cause of partial feedwater isolation was inadvertent actuation of a relay in the safety injection actuation circuitry during logic testing. Actuation of this relay caused the tripping of the power supply to one of the Main Feedwater (MFW) pump motors and the closure of the MFW pump's discharge isolation valve. The Reactor Protection System functioned as designed, and post-trip response was satisfactory. The reactor was placed in a safe, hot shutdown condition. The health and safety of the public were not affected. This report is being made pursuant to 10CFR50.73(a)(2)(iv).

END OF ABSTRACT

ATTACHMENT TO 9406160107 PAGE 1 OF 2

Chart "REQUIRED NUMBER OF DIGITS/CHARACTERS FOR EACH BLOCK" omitted.

TEXT PAGE 2 OF 4

1.0 DESCRIPTION OF THE EVENT

On May 11, 1994, with Unit 1 initially at 100% power the reactor was manually tripped at 2109 hours. Immediately before the trip, at 2108 hours, an alarm was received in the Main Control Room indicating that circuit breaker 15C5, which supplies one of the Main Feedwater Pump B tandem drive motors EIIS:SJ,MO!, had opened. Because of the supply breaker-feedwater discharge valve interlock scheme, the associated discharge valve, 1-FW-MOV-154B EIIS:SJ,ISV! was closing. Control Room Operators immediately began reducing turbine load and reactor power in an effort to restore stable conditions. Reduction in feedwater supply and the shrink from power reduction caused steam generator levels to continue to decrease, and the Control Room Operator manually tripped the reactor.

All control rods fully inserted into the core, and the turbine EIIS:TA! and main generator EIIS:TB! tripped as designed. The Anticipated Transient Without Scram Mitigation System Actuation Circuitry (AMSAC) also actuated as designed. The Auxiliary Feedwater (AFW) Pumps EIIS:BA,P! 1-FW-P-2, 1-FW-P-3A and 1-FW-P-3B, started automatically and supplied feedwater to the steam generators EIIS:AB,SG! as designed.

The main steam dumps EIIS:SB,V! automatically opened to admit steam directly to the main condenser EIIS:COND!. The Reactor Coolant System (RCS) EIIS:AB! average temperature (Tave) was reduced to 547 degrees Fahrenheit and the main steam dumps closed as designed. RCS Tave decreased to a minimum of approximately 535 degrees Fahrenheit. RCS temperature subsequently stabilized at 547 degrees Fahrenheit (no-load temperature).

Intermediate Range Nuclear Instrument (IRNI) EIIS:IG,JI! N-36 indicated off-scale low.

Control Room Operators responded to the trip in accordance with emergency and other operating procedures. Plant response was as expected.

A four-hour non-emergency report was made to the Nuclear Regulatory

Commission in accordance with 10CFR50.72(b)(2)(ii) at 2348. This event is being reported pursuant to 10CFR50.73(a)(2)(iv), manual actuation of the Reactor Protection System (RPS) EIIS:JC!.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

When the reactor was manually tripped, RPS actuations functioned as designed and all control rods fully inserted into the core. The electrical buses transferred properly, and offsite power was maintained throughout the event. The emergency diesel generators (EDG) EIIS:EK,DG! remained operable in automatic, but were not required to start. The AFW Pumps started and supplied feedwater to the steam generators as designed. Station operating personnel acted promptly to place the plant in a stable, hot shutdown condition. The shutdown margin of reactivity was calculated and found to be satisfactory. No conditions adverse to safety resuLted from this event, and the health and safety of the public were not affected.

3.0 CAUSE OF THE EVENT

Just before the reactor trip, Licensee Instrument and Calibration (I&C) Technicians were conducting surveillance testing in accordance with procedure 1-IPT-FT-RP-SI-001A, Train A Safeguards Actuation Logic Testing. A step in this procedure requires the measurement of electrical resistance of the relay coil F1A EIIS:JE,RLY,CL! in the actuation circuitry of Safety Injection Train A. This relay is located in the A Train Protection Rack with restricted access to the terminals. When the technician inserted the ohmmeter probes to take the measurement, he inadvertently disturbed the contacts which control the trip coil for the MFW pump motor EIIS:SJ,MO,CL(94)! as described in Section 1.0 above. The circuit breaker tripped, initiating the sequence of events which led to the manual reactor trip. The cause of the event was human error; however, the difficulty in testing the affected relay was a major contributing factor.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Following the trip, Control Room Operators acted promptly to place the plant in a safe, hot shutdown condition in accordance with emergency and other operating procedures. The Shift Technical Advisor monitored the critical safety function status trees to verify that unit conditions were acceptable. Plant response was as expected, and the unit was stabilized at hot shutdown.

5.0 ADDITIONAL CORRECTIVE ACTION(S)

Intermediate Range Nuclear Instrument (IRNI) N-36 indicated off-scale low. This IRNI response has been observed following previous reactor trips and evaluated. The overcompensated response following a reactor trip is considered normal and is the expected response that may result when using the NSSS suppliers preferred methodology for setting compensating voltage. The Source Range Nuclear Instruments (SRNI) automatically reinstated as designed.

When the main steam dumps automatically opened to admit steam directly to the main condenser, the Reactor Coolant System (RCS) average temperature (Tave) was reduced to 547 degrees Fahrenheit and the main steam dumps closed as designed. The RCS cooldown minimum temperature of approximately 535 degrees Fahrenheit is expected and has been observed during previous reactor trips. RCS temperature subsequently stabilized at 547 degrees Fahrenheit (no-load temperature). This condition has been evaluated previously. Corrective actions are being taken based on the results of this evaluation. Shutdown margin calculations following a reactor trip include a conservatively safe value for RCS temperature ensuring adequate shutdown margin exists.

A post trip review was conducted and no additional corrective actions were identified.

6.0 ACTIONS TO PREVENT RECURRENCE

Although the major cause of the transient was human error, a significant contributor was the inaccessibility of the circuit being tested. The following corrective actions are underway:

o A plant modification has been developed which will locate test points in positions where required test data may be obtained without the risk of inadvertent actuation. It is anticipated that this modification will be installed during the next refueling outage for each unit.

o An evaluation is being conducted to determine if the interval between the required surveillance testing of this and similar relays may be increased without adversely affecting nuclear safety. If such a change is permissible, the number of potential challenges to plant stability will be significantly reduced.

7.0 SIMILAR EVENTS

LER S2-90-004, Unit 2 Manual Reactor Trip Following Inadvertent Grounding During Testing.

8.0 MANUFACTURER/MODEL NUMBER

None.

ATTACHMENT TO 9406160107 PAGE 1 OF 1

10CFR50.73

Virginia Electric and Power Company
Surry Power Station
P
O. Box 315
Surry, Virginia 23883

June 9, 1994

U.S. Nuclear Regulatory Commission Serial No.: 94-360
Document Control Desk SPS:MDK
Washington, D.C. 20555 Docket No.: 50-280
License No.: DPR-32

Dear Sirs:

Pursuant to Surry Power Station Technical Specifications, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Unit 1.

REPORT NUMBER

50-280/94-006-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

M. R. Kansler
Station Manager

Enclosure

cc: Regional Administrator
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

*** END OF DOCUMENT ***